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TIME DEPENDING ASSESSMENT OF LOW AND INTERMEDIATE RADIOACTIVE WASTE CHARACTERISTICS FROM CERNAVODA NPP

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ABSTRACT

Low and intermediate radioactive gaseous, liquid and solid waste produced at Cernavoda Nuclear Power Plant must be well known from the point of view of contained radionuclide activity, during all steps of their processing, storage and transport, to ensure the nuclear safety of radioactive waste management.

As in intermediate storage stage, the waste activity changes by radioactive decay and nuclear transmutation, the evolution in time of these sources is necessary to be assessed, for the purpose of biological shielding determination at any time.

On the other hand, during the transport of waste package at the repository, the external dose rates must meet the national and international requirements concerning radioactive materials transportation on public roads.

In this paper, a calculation methodology for waste characterization based on external exposure rate measurement and on sample analysis results is presented. The time evolution of waste activity, as well as the corresponding shielding at different moments of management process, have been performed using MICROSHIELD-5 code.

The spent resins proceeded from clean-up and purification systems and solutions from decontamination have been analyzed. The proposed methodology helps us to assess radiation protection during the handling of low and intermediate - level radioactive waste drums, ensuring safety conditions for the public and environment.

1 INTRODUCTION

Low and intermediate radioactive gaseous, liquid and solid waste produced at Cernavoda Nuclear Power Plant must be well known from the point of view of contained radionuclide activity, during all steps of their processing, storage and transport, to ensure the nuclear safety of radioactive waste management.

The gaseous wastes are filtered and released under control through the ventilation stack, therefore the paper takes into consideration only liquid and solid wastes.

As the waste activity changes in time by radioactive decay and nuclear transmutations, it is very important to know the evolution in time of these sources, in order to adopt the most adequate measures for radiological protection, including radiation shielding, at any moment.

On the other hand, during the transport of waste package at the repository, the external dose rates must meet the national and international requirements concerning radioactive materials transportation on public roads.

In this paper a calculation methodology is presented for characterization of the spent ionic resins and the solutions resulted from decontamination, based on external exposure rate measurement and on sample analysis, using the computer code MICROSHIELD-5 [1].

2 WASTE DESCRIPTION

2.1 Liquid wastes

The radioactive liquid wastes are classified into aqueous and organic liquid wastes.

The aqueous liquid wastes are collected by the radioactive liquid waste management system, and are of the following types:

- Level 1 radioactive wastes – resulting from laundry, shower rooms, labs and floor drainage in the Service Building, called low – level wastes, with an average radioactivity of 1.85×10^2 Bq/l;
- Level 2 radioactive wastes – resulting from the heavy water upgrading system, equipment decontamination, laundry for plastic objects, labs and floor drainage in the Service Building, called intermediate – level wastes, with an average radioactivity of 1.85×10^4 Bq/l;
- Level 3 radioactive wastes – resulting mainly from the Reactor Building drainage system, spent fuel storage bay, spent resin storage vaults, with an average radioactivity of 1.85×10^6 Bq/l; normally, these wastes are collected together with the Level 2 wastes, the resulting mixture having an average radioactivity of 1.85×10^5 Bq/l.

2.2 Solid wastes

During the NPP operation, the following types of solid radioactive wastes are produced:

- Low activity radioactive waste – Type 1 (gamma dose rates of less than 2 mGy/h on contact with the container containing the wastes);
- Medium activity radioactive waste – Type 2 (gamma dose rates of 2 mGy/h to 125 mGy/h on contact with the container);
- Medium activity radioactive waste – Type 3 (gamma dose rates in excess of 125 mGy/h on contact with the container).

The spent ionic resins are obtained from different purification systems: of primary coolant, moderator, water from spent fuel bay, etc., having a dose rate on contact greater than 10^{-2} Gy/h. Therefore, special protection measures, including shielding, must be taken to transport, handle and store them.

The activity of the resins from the primary coolant purification system or the spent fuel bay water, is mainly due to the isotopes ^{134}Cs and ^{137}Cs present in the fuel bundles; the activity produced in the moderator purification system is mainly due to the isotopes ^{60}Co and ^{51}Cr, resulted from the neutron activation of the structural materials.

Spent resins are stored for 10 years in concrete vaults in the Service Building basement, and then are conditioned by cementing in standard metallic drums of 200 l.

3 COMPUTING STEPS

MICROSHIELD-5 is a multi-group 3-D computer code and represents an improved version of MICROSHIELD-4.

MICROSHIELD-5 estimates by point kernel as well as analytical methods, photonic and energetic fluence, gamma exposure, absorbed gamma dose rate in air and dose equivalent, for different geometrical models of calculation.

The code possibilities regarding gamma exposure estimation, include:

- designing of the shields and cask walls;
- evaluation of the radiation exposure for humans and materials;
- selection of the temporary shields during maintenance operations;
- evaluation of the source intensity from the dose measurements at the waste storage facilities;
- minimizing of the person exposures.

Typical steps of calculation for waste assessment, corresponding to their handling, storage and transport are:

- activity inference at different times, based on the external dose measured at the surface of the waste container, the radionuclide concentration resulted from the waste sampling (before cementation), and/or the volumetric cementation ratio;
- shield (permanent or temporary) and external exposure rate evaluation (using the source previously calculated) during storage of a drum containing solidified wastes, as well as transportation at the low and intermediate-level waste final repository.

The evaluation takes into account the decay of the radionuclides contained in the waste package.

In this context, the code MICROSHIELD-5 uses the following menu facilities:

1. "external source file" (a file containing the external source, representing the results from sample measurement, is created);
2. "source inference" - the total activity of the source, based on actual exposure measurements, is inferred;
3. multiple dose points (one to four dose points can be defined);
4. "exposure rate vs. decay time" (exposure rate variation with decay time is performed);
5. "sensitivity" (variation of interest parameters: dose point location, shield thickness, source size and orientation vs. dose point location, quadrature order, can be used).

4 APPLICATIONS

Three examples significant for waste characterization are described below [2].

4.1 Example 1 - Drum filled with resins

Assume a container filled with dewatered resins, with the following composition resulted from sample analysis at the moment t_1:

Co – 60	0.25 µCi/cm^3
Cs – 134	1.25 µCi/cm^3
Cs – 137	1.50 µCi/cm^3
Sb – 125	0.44 µCi/cm^3

At the moment t_2 the resins were transferred into a drum and cemented; thus, the drum contains dewatered resins and concrete in an unknown ratio.

Geometrical model for MICROSHIELD is presented in Figure 1.

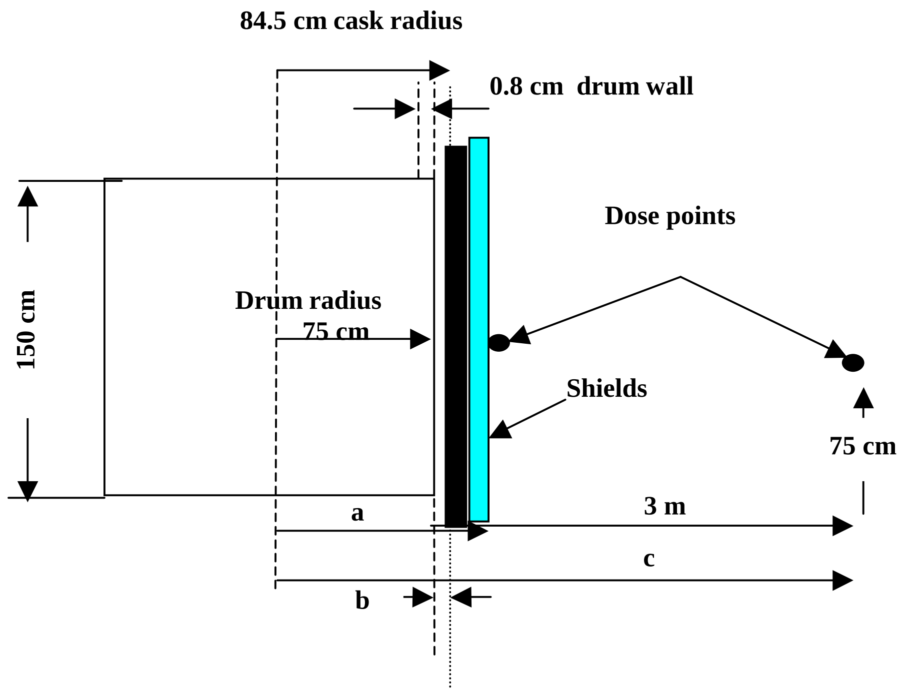

Figure 1: Model for a drum filled with resins

Immediately after solidification (at the moment t_2) the measured dose rate on the drum surface was 1.294 mGy/h.

The drum is stored for a period and, at the moment t_3, will be transported (in a transport cask) to the final repository.

The following characteristics have been calculated:

- total activity of the drum calculated from a dose measurement, considering or not the radiation buildup (Table 1);
- dose rate on the transport cask surface (Table 2);
- shield thickness necessary to meet the requirements regarding the dose limitation during transportation.

It was concluded that a shield consisting in 6 cm of lead and 6 cm of steel ensures a gamma dose rate at 1 m from the cask surface, of 82 μGy/h.

Table 1 Activity of the drum calculated from a dose measurement (Ci)

Radionuclide	At the moment t_2 (49 days decay)		At the moment t_3 (100 days decay)	
	Without buildup	With buildup	Without buildup	With buildup
Ba-137m	12.75	3.870	11.86	3.600
Co-60	2.210	0.670	1.460	0.440
Cs-134	10.77	3.270	3.750	1.140
Cs-137	13.48	4.090	12.54	3.800
Sb-125	3.830	1.160	1.750	0.530
Te-125m	0.400	0.120	0.430	0.130
Total	43.44	13.17	31.79	9.640

Table 2 Dose rate on the transport cask surface (μGy/h)

Dose point	Without buildup	With buildup
Contact on cask surface	10.2	150.9
1 m from cask surface	5.8	82.0
2 m from cask surface	2.9	40.8

4.2 Example 2 – Container filled with cemented aqueous solution

An aqueous solution from decontamination has been neutralized, concentrated and solidified with cement. The dilution ratio is unknown.

The solidification container is cylindrical, made of steel, 1 meter high, 1 meter in diameter, and 2 cm thick.

The solidified material will be placed in the storage facility for at least 2 years before transportation.

A sample of the solution was analyzed 10 days before a dose measurement (before solidification).

Exposure rate measured at 1 meter from the side and halfway up the height of the container, after solidification, was 10.29 mGy/h.

The results of the sample analysis are:

Co – 60	15.1 µCi/cm^3
Cr – 51	5.4 µCi/cm^3
Fe – 59	7.88 µCi/cm^3
Mn – 54	2.13 µCi/cm^3
Ni – 59	0.125 µCi/cm^3

The geometrical model used is similar with Figure 1.

The following characteristics have been calculated:

- total activity in the solidified material at 10 days and 2 years (Table 3);
- external exposure dose rate during transport at 1 m from the container and halfway up the height (Table 4).

Table 3 Activity of the cemented waste

Radio-nuclide	Infered source activity – with buildup (Ci)	
	10 days	2 years
Co-60	12.20	9.41
Cr-51	3.41	4.99 E-8
Fe-59	5.47	7.50 E-5
Mn-54	1.69	0.34
Ni-59	0.10	0.10
Total	22.87	9.85

Table 4 Exposure rate at 1 m from the container during transport

Decay time (years)	Exposure rate – with buildup (mGy/h)
0	10.29
2	6.389

4.3 Example 3 – Container filled with resins placed in a storage building

The example presented in Figure 3 consists of a container filled with resins, placed in a radwaste storage building.

Concentrations in a resin sample from the demineralizer, taken when the resins are removed, but before they are placed in container, are following:

Co – 58	28.13 μCi/cm^3
Co – 60	40.50 μCi/cm^3
Cs – 137	187.50 μCi/cm^3
Mn – 54	15.63 μCi/cm^3
Sb – 125	14.13 μCi/cm^3

After sampling and analysis (considered as reference moment), the resins are solidified with cement at a volumetric ratio of 20% resin.

The external measurement made 2 weeks after the reference date, at 15 cm distance from the surface of the container, was 562 mGy/h.

The following characteristics have been estimated:

- waste activity deduced from the sample analysis and dose measurement;
- external exposure dose rate at 15 cm from surface of the container, after 10 years, when the resins will be placed in a final repository (241 μGy/h);
- size of the shields necessary for transportation of the waste container to the repository;
- the shielding necessary for waste disposal.

Finally, it resulted that for a shield consisting of 35 cm of concrete (building wall) and 6 cm of steel (portable shield) the following values were obtained: 92.3 μGy/h on surface and 51.6 μGy/h at 1 m from surface.

Comparing the results with the dose limits applicable for the disposal facility, it may be concluded that the shield above mentioned needs to be supplemented, in order to meet the regulation requirements.

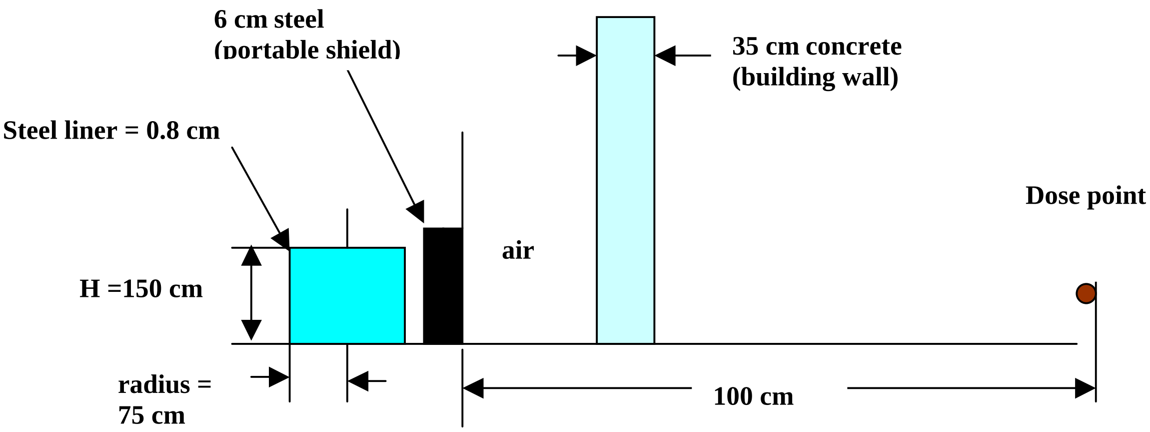

Figure 2: Model of a container placed in a storage building

5 CONCLUSIONS

The calculation methodology proposed in this paper related to the characterization of the low and intermediate – level radioactive wastes, could be applied to the management of the wastes resulted from Cernavoda NPP and/or from other nuclear facilities, to estimate the time evolution of waste activity, and the necessary radiation shielding at any moment of waste management, including the transportation to the final repository.

6 REFERENCES

[1] MICROSHIELD – Version 5, User's Manual, Grove Engineering, 1998

[2] S. Mateescu, M. Stanciu, *Variation of the Microshield input parameters for characterization of the radioactive waste from Cernavoda NPP*, Internal Report, 2002